ensure that the onsite storage of spent fuel is in compliance with GDC 62 for the prevention of criticality in fuel storage and handling and with the 5 percent subcriticality margin position of the NRC staff to assure compliance with GDC 62.

Required Response

All addressees are required to submit a written response to the information requested above within 120 days of the date of this generic letter. If an addressee chooses not to respond to specific questions, an explanation of the reason and a description of any proposed alternative course of action should be provided, as well as the schedule for completing the alternative course of action (if applicable), and the safety basis for determining the acceptability of the planned alternative course of action.

Address the required written reports to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, submit a copy to the appropriate regional administrator.

Backfit Discussion

This generic letter only requires information from the addresses under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). Therefore, the staff has not performed a backfit analysis. The information requested will enable the NRC staff to determine whether licensees are complying with the current licensing basis for the facility with respect to GDC 62 for the prevention of criticality in fuel storage and handling and 5 percent subcriticality margins either contained in the technical specifications, or committed to in the updated FSARs, of plants containing Boraflex in the spent fuel storage racks. The staff is not establishing a new position for such compliance in this generic letter. Therefore, this generic letter does not constitute a backfit and no documented evaluation or backfit analysis need be prepared.

Federal Register Notification

(To be completed after the public comment period.)

Paperwork Reduction Act Statement

The information collections contained in this request are covered by the Office of Management and Budget clearance number 3150–0011, which expires July 31, 1997. The public reporting burden for this collection of information is estimated to average 150 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch, (T–6F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB–10202 (3150–0011), Office of Management and Budget, Washington, DC 20503.

Dated at Rockville, Maryland, this 2nd day of November, 1995.

For the Nuclear Regulatory Commission.

Dennis M. Crutchfield,

Director, Division of Reactor Program Management, Office of Nuclear Reactor Regulation.

[FR Doc. 95–27624 Filed 11–7–95; 8:45 am]

BILLING CODE 7590–01–P

Issuance of Urgent Bulletin; NRC Bulletin 95–02, Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of issuance.

SUMMARY: The Nuclear Regulatory Commission (NRC) has issued Bulletin 95–02 to request certain remedial actions and associated reporting by holders of boiling water reactor (BWR) licenses and construction permits as a result of the unexpected clogging of a residual heat removal pump strainer at a boiling water reactor facility while operating in the suppression pool cooling mode. This bulletin is available in the NRC Public Document Room under accession number 9510040059. This bulletin was issued as an urgent generic communication under NRC procedures for issues that the staff considers urgent. This bulletin is discussed in Commission information paper SECY–95–255 which is also available in the NRC Public Document Room.

DATES: The bulletin was issued on October 17, 1995.

ADDRESSES: Not applicable.

FOR FURTHER INFORMATION CONTACT: Robert B. Elliott, (301) 415–1397 or Robert M. Latta, (301) 415–1314.

SUPPLEMENTARY INFORMATION: The NRC issued this bulletin to accomplish the following:

(1) Alert BWR owners to complications experienced during a recent event in which a licensee initiated suppression pool cooling in response to a stuck-open safety relief valve (SRV) and subsequently experienced clogging of one RHR pump suction strainer.

(2) Request BWR owners to review the operability of their emergency core cooling system (ECCS) and other pumps which draw suction from the suppression pool while performing their safety function. The evaluation should be based on suppression pool cleanliness, suction strainer cleanliness, and the effectiveness of foreign material exclusion (FME) practices. In addition, BWR owners are requested to implement appropriate procedural modifications and other actions (e.g., suppression pool cleaning), as necessary, to minimize foreign material in the suppression pool, drywell and containment. BWR owners are requested to verify their operability evaluation through appropriate testing and inspection.

(3) Require that BWR owners report to the NRC whether and to what extent they have complied with the requested actions. In addition, require a second report indicating completion of confirmatory test(s) and inspection(s) and providing the test results by BWR owners that have complied with the requested actions, or indicating completion of any proposed alternative course of action by BWR owners that have not complied with the requested actions.

Dated at Rockville, Maryland, this 2nd day of November, 1995.

For the Nuclear Regulatory Commission.

Dennis M. Crutchfield,

Director, Division of Reactor Program Management Office of Nuclear Reactor Regulation.

[FR Doc. 95–27625 Filed 11–7–95; 8:45 am]

BILLING CODE 7590–01–P

Biweekly Notice

Applications and Amendments to Facility Operating LicensesInvolving No Significant Hazards Considerations

I. Background

Pursuant to Public Law 97-415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97-415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from October 14, 1995, through October 27, 1995. The last biweekly notice was published on Wednesday, October 25, 1995 (60 FR 54714).

Notice Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the Federal Register a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this Federal Register notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By December 8, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

Arizona Public Service Company, et al., Docket Nos. STN 50-529 and STN 50-530, Palo Verde Nuclear Generating Station, Units Nos. 2 and 3, Maricopa County, Arizona

Date of amendments request: October 3, 1995

Description of amendments request: The amendment would delete the provisions relating to certain previous sale and leaseback transactions that were by added by Amendment No. 3 for NPF-51 and Amendment No. 1 for NPF-74.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

This amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed change is administrative in nature. The proposed change deletes Sections 2.B.(7)(a) and (b) of License No. NPF-51, and Sections 2.B.(6)(a) and (b) of License No. NPF-74. These sections describe the structure of the financing of El Paso's interest in Palo Verde, specifically authorizing sale and leaseback transactions. The proposed change does not affect the assumptions used in the accident analyses, nor does the proposed change result in changes to the physical configuration of the facility, design parameters, technical specifications, or operation and maintenance of the facility. Therefore, the amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

This amendment request does not create the possibility of a new or different kind of accident from any accident previously analyzed because the proposed change is administrative in nature. The proposed change deletes Sections 2.B.(7)(a) and (b) of License No. NPF-51, and Sections 2.B.(6)(a) and (b) of License No. NPF-74. These sections describe the structure of the financing of El Paso's interest in Palo Verde Units 2 and 3, specifically authorizing sale and leaseback transitions. The proposed change does not involve modifications to any of the existing equipment nor does the change affect operation or maintenance of the facility. Therefore, the amendment request does not create the possibility of a new or different kind of accident not previously analyzed.

3. The proposed change does not involve a significant reduction in a margin of safety.

This amendment request does not involve a significant reduction in a margin of safety because it is administrative in nature. The proposed change deletes Sections 2.B.(7)(a) and (b) of License No. NPF-51, and Sections 2.B.(6)(a) and (b) of License No. NPF-74. These sections describe the structure of the financing of El Paso's interest in Palo Verde, specifically authorizing the sale and leaseback transactions. The proposed change does not involve changes to any existing plant equipment or accident analyses that provide for or establish margins of safety. There is no change to the operation or maintenance of the facility and the existing margins of safety are not changed by the proposed change. Therefore, the amendment request does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on that review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involve no significant hazards consideration.

Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004

Attorney for licensee: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072-3999

NRC Project Director: William H. Bateman

Baltimore Gas and Electric Company, Docket No. 50-318, Calvert Cliffs Nuclear Power Plant, Unit No. 2, Calvert County, Maryland

Date of amendment request: October 2, 1995

Description of amendment request: The proposed amendment would revise the Calvert Cliffs Nuclear Power Plant, Unit No. 2, Technical Specifications on a one-time basis by increasing the 7 day allowed outage time (AOT) of the control room emergency ventilation system (CREVS) to an AOT of 30 days. This requested one-time increase in the AOT is applicable only for the loss of the emergency power supply to one train of the CREVS during the Unit No. 1 spring 1996 refueling outage.

The requested extension in the AOT is necessary to allow the licensee to perform modifications to the electrical distribution system during the upcoming Unit 1 refueling outage while Unit No. 2 continues to operate. The modifications include connecting a fourth safety-related (SR) emergency diesel generator (EDG) to engineered safety features (ESF) Bus No. 11. The work related to this effort will require that the bus be deenergized for several days isolating it from its normal and emergency EDG power supplies. One train of the CREVS is connected to ESF Bus No. 11 and will not have its power supplies available for a period of time. The normal (offsite) power is expected to be restored in about 3 days, but the emergency power (onsite EDG) may take up to 30 days.

The licensee is taking additional actions to assure the availability of the normal offsite power source and is also adding a nonsafety-related (NSR) EDG as an alternate onsite power source during the period that the SR EDG is not available. The licensee expects that the tie-in of the NSR EDG will take about 8 days. Thus, even if the normal offsite power source is lost, the temporary onsite NSR EDG will be available to provide power to the affected train of the CREVS.

Basis for proposed no significant hazards consideration determination: As required by 10CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Control Room Emergency Ventilation System (CREVS) is used to mitigate the consequences of an accident. It is designed so that the Control Room remains habitable for operators and to maintain the environment needed for continued equipment operation. The system is redundant (two 100% capacity trains) and is powered from both normal (offsite) and emergency (emergency diesel generators) power sources. We [the licensee] are proposing an amendment which would allow the emergency power to be removed from one of the redundant CREVS for an additional 23 days (beyond the 7 days allowed by the Technical Specifications). Other than the removal of the emergency electrical power source, we are not affecting or modifying the operation of the CREVS. The CREVS is not an accident initiator for any previously evaluated accident. Therefore, the proposed change does not involve an increase in the probability of an accident previously evaluated.

The CREVS is designed to mitigate the consequences of design basis accidents. For that purpose, redundant trains are provided to protect against a single failure. During the Technical Specification seven day Allowed Outage Time (AOT), an operating unit is allowed by the Technical Specifications to remove one of the CREVS trains from service, thereby eliminating this single failure protection. The consequences of a design basis accident coincident with a failure of the redundant CREVS train during the additional 23-day period are the same as those during the 7-day AOT. Therefore, the proposed change does not significantly increase the consequences of an accident previously evaluated.

Therefore, the proposed change does not increase the probability or consequences of an accident previously evaluated.

2. Would not create the possibility of a new or different type of accident from any accident previously evaluated.

The CREVS is not being modified by this proposed change nor will any unusual operator actions be required. The system will continue to operate in the same manner. The CREVS is not an initiator to any accident, but is designed to respond should an accident occur.

Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Would not involve a significant reduction in a margin of safety.

The operability of the CREVS during Modes 1 through 4 ensures that the Control Room will remain habitable for operators and to maintain the environment needed for continued equipment operation under all plant conditions. The proposed change does not affect the function of the CREVS. During the period of the Technical Specifications AOT when one CREVS train is inoperable, the margin of safety is reduced. This time period is a temporary relaxation of the single failure criteria, which, consistent with overall system reliability considerations, provides a limited time to maintain or repair the equipment and conduct testing. We are requesting an extension of this limited time. The proposed change will allow one train of the CREVS to be without an emergency power supply for an additional 23 days beyond the 7-day AOT (total of 30 days). This train of CREVS will be functional and will have the normal power supply available for all but approximately three days to allow work and necessary testing on the bus. The other train of the CREVS will have both its normal and emergency power supplies during this period.

To provide additional assurance that all reasonable steps have been taken to prevent the loss of the normal power supply to the CREVS, we will restrict maintenance activities on three of the four offsite transmission lines. This restriction will cover the period we are in the Action Statement for the CREVS (Action Statement 3.7.6.1.a and b). To provide an alternative power source during the majority of this period, we will connect the Alternate AC power source (No. 0C Diesel Generator) to ESF Bus No. 11 and confirm its availability as soon as possible after the work on ESF Bus No. 11 begins (we [the licensee] expect that to take about eight days). This power source is independent from the offsite power supplies. In addition, we will restrict planned maintenance on the No. 12 CREVS during the period we are in the Action Statement to ensure that the No. 12 CREVS is not removed from service.

We believe that the reduction in the margin of safety represented by this one-time extension of the AOT is not significant based on our management of plant risk, the reliability of the normal CREVS power supply, the availability of the redundant CREVS with both its normal and emergency power, and the mitigating features described above. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678

Attorney for licensee: Jay E. Silbert, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Project Director: Ledyard B. Marsh

Carolina Power & Light Company, et al., Docket Nos. 50-325 and 50-324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of amendments request: October 23, 1995

Description of amendments request: The amendments would delete the applicability of the primary coolant water chemistry limits when the primary system is being chemically decontaminated and the reactor vessel is defueled.

Basis for proposed no significant hazards consideration determination: As required by 10CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed changes will allow the reactor coolant system conductivity and chlorides to exceed the limits specified in Technical Specification Table 3.4.4-1 in support of performing chemical decontamination activities. The reactor coolant system water chemistry limits have been established to prevent long-term damage to the reactor coolant system materials that are in contact with the coolant. Upon concluding the chemical decontamination activities, reactor coolant system conductivity and chloride values would be restored to within the limits specified in Technical Specification Table 3.4.4-1. Existing regulatory requirements, specifically a review in accordance with 10 CFR 50.59 to determine whether an activity involves an unreviewed safety question, provide adequate assurance that solvents selected for use in a chemical decontamination activity will not degrade the structural integrity of the reactor coolant system. Therefore, since the structural integrity of the reactor coolant system will not be adversely impacted by the chemical decontamination activities, the proposed amendments do not involve a significant increase in the probability of an accident previously evaluated.

As discussed above, the reactor coolant system water chemistry limits have been established to prevent long-term damage to the reactor coolant system materials that are in contact with the coolant. The solvents being used for a chemical decontamination activity are selected to ensure their effectiveness and to ensure that damage will not occur to the structural materials comprising the reactor coolant pressure boundary. As such, the operation of safety equipment used to mitigate a design basis accident or transient will not be affected by the proposed change of the reactor coolant system water chemistry limits during performance of chemical decontamination activities. Therefore, the proposed revision to the reactor coolant system chemistry limits will not involve a significant increase in the consequences of an accident previously evaluated.

2. The proposed change will allow the reactor coolant system conductivity and chlorides to exceed the limits specified in Technical Specification Table 3.4.4-1 in order to perform chemical decontamination activities. The reactor coolant system water chemistry limits have been established to prevent long-term damage to the reactor coolant system materials that are in contact with the coolant. Even though the solvents used for chemical decontaminations may result in reactor coolant system conductivity and chloride measurement values in excess of the limits specified in the Technical Specifications, the existing regulatory requirements of 10 CFR 50.59 will continue to ensure that solvents being used for performing chemical decontamination have been properly evaluated and that these solvents do not adversely affect the material properties or structural integrity of the reactor coolant system. Therefore, the proposed amendments revising the reactor coolant system water chemistry limits during performance of chemical decontamination activities will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The reactor coolant system water chemistry limits have been established to prevent long-term damage to the reactor coolant system materials that are in contact with the coolant. The solvents used for chemical decontaminations result in reactor coolant system conductivity and chloride measurement values in excess of the limits specified in the Technical Specifications; however, the solvents being used of performing chemical decontamination have been properly evaluated to ensure they will not significantly affect the material properties of the reactor coolant system piping (i.e., corrosion) nor will they significantly affect the structural integrity (i.e., wall thinning) of the reactor coolant system piping. Therefore, the proposed license amendments do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297

Attorney for licensee: R. E. Jones, General Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: David B. Matthews

Consolidated Edison Company of New York, Docket No. 50-247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of amendment request: November 2, 1994, as supplemented on January 4, 1995

Description of amendment request: The amendment would revise the Technical Specifications (TSs) to make editorial changes, delete portions of the TSs that have become unnecessary due to previously approved amendments, change managerial titles, update references and reporting requirements, revise the Station Nuclear Safety Committee (SNSC) composition to specify disciplines rather than specific job titles, modify the record keeping requirements of the Nuclear Facilities Safety Committee, implement changes referenced in Generic Letter 93-07, ''Modification of the Technical Specification Administrative Control Requirements for Emergency and Security Plans,'' and to correct the shift manning requirements table.

Basis for proposed no significant hazards consideration determination: As required by 10CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. There is no significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments are administrative in nature. They involve making editorial changes, deleting portions of the Technical Specifications that have become unnecessary due to previously approved amendments, changing managerial titles, updating references and reporting requirements, revising the SNSC composition to specify disciplines rather than specific job titles, implementing changes referenced in Generic Letter 93-07, and revising shift manning to conform with the requirements of 10 CFR 50.54. These changes do not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety Systems Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed changes to the subject Technical Specifications would not increase the probability or consequences of an accident previously evaluated.

2. The possibility of a new or different kind of accident from any accident previously evaluated has not been created.

As stated above, the proposed changes are administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently, no new failure modes are introduced as a result of the proposed changes. Therefore, the proposed changes would not initiate any new or different kind of accident.

3. There has been no significant reduction in the margin of safety.

The proposed changes are administrative in nature. Since there are no changes to the physical design or operation of the facility, the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed changes would not result in a reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610

Attorney for licensee: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003

NRC Project Director: Ledyard B. Marsh

Consolidated Edison Company of New York, Docket No. 50-247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of amendment request: August 29, 1995

Description of amendment request: The proposed amendment would revise Technical Specification Sections 3.1.F and 4.13 to provide for appropriate inservice inspection for any steam generator tubes containing sleeves and to provide for reduced allowable primary-to-secondary leakage rates for steam generators containing sleeves. The proposed changes are in response to commitments made by Consolidated Edison by letter dated April 5, 1995, during the review of an amendment which permitted the use of laser welded steam generator tube sleeves as a method of tube repair.

Basis for proposed no significant hazards consideration determination: As required by 10CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification Amendment No. 183 allowed sleeving as an acceptable alternate tube repair method for Indian Point Unit No. 2. The steam generator sleeve approved for installation is the Westinghouse process (laser welded sleeve). The sleeve configuration was designed and analyzed in accordance with the criteria of Regulatory Guide (RG) 1.121 and the design requirements of Section III of the American Society of Mechanical Engineers (ASME) Code. Fatigue and stress analyses of the sleeved tube assembly produced acceptable results as documented in the Westinghouse topical report submitted in the original sleeving package. Mechanical testing has shown that the structural strength of the sleeves under normal, faulted, and upset conditions is within acceptable limits. Leakage rate testing for the tube sleeves has demonstrated that primary-to-secondary leakage is not expected during all plant conditions.

Any leakage through the sleeved region of the tube is fully bounded by the leak-before-break considerations and, ultimately, the existing steam generator tube rupture analysis included in the Updated Final Safety Analysis Report (UFSAR).

The reduction in TS leakage rate requirements from 0.3 gpm [gallons per minute] (432 gpd [gallons per day]) allowable per SG to 150 gpd per steam generator containing sleeves further ensures that SG tube integrity is maintained in the event of a main steam line break (MSLB) or under Loss Of Coolant Accident (LOCA) conditions. The RG 1.121 criteria for establishing operational leakage rate limits require a plant shutdown based upon a leak-before-break consideration to detect a free span crack before a potential tube rupture. The 150 gpd limit will continue to allow for early leakage detection and require a plant shutdown in the event of tube leakage that exceeds the revised Technical Specification limit.

The sleeve sample size has been increased to a minimum of twenty (20) percent of the inservice sleeves. Increasing the sample size of the sleeves to be inspected will increase the monitoring of tubes using sleeves for any further degradation while they remain inservice. If the sample identifies a sleeve with an imperfection of greater than 23 percent depth an additional 20 percent of the sleeves shall be inspected. The sleeves that have identified imperfections of greater than 23 percent shall be evaluated and removed from service.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Implementation of the proposed amendment will not introduce significant or adverse changes to the plant design basis. The proposed changes do not involve plant modification or changes to equipment, and consist of reducing the allowable steam generator leakage limits for steam generators containing sleeves and defining the sample size of the steam generator tube sleeve inspection.

The reduction in TS leakage rate requirements from 0.3 gpm (432 gpd) allowable per SG to 150 gpd per SG containing sleeves further ensures that SG tube integrity is maintained in the event of a MSLB or under LOCA conditions. The 150 gpd limit is designed to provide for leakage detection and a plant shutdown in the event of the concurrence of excessive tube leakage. The limit provides for early detection and a plant shutdown prior to a postulated defect reaching critical magnitudes for Main Steam Line Break conditions.

Formalizing the sample size of sleeved tubes inspected during each scheduled inservice inspection will ensure increased monitoring of these tubes for any further degradation. The improved monitoring and evaluation of the tube and the sleeves assures tube structural integrity is maintained or the tube is removed from service.

With these actions the possibility of a new or different type of accident from any accident previously evaluated is not created.

3. The proposed change does not involve a significant reduction in a margin of safety.

Implementation of the proposed changes will not reduce the margin of safety. This amendment involves the reduction of sleeved steam generator tube leakage limit and a formalized inservice inspection program for sleeved tubes. These actions will help ensure steam generator tube integrity.

Reduction of the leakage rate requirement from 0.3 gpm (432 gpd) to 150 gallons per day (gpd) per sleeved steam generator will continue to ensure steam generator tube integrity is maintained in the event of main steam line break or under LOCA conditions. Reducing this limit will not result in a reduction in the margin of safety.

The portions of the installed sleeve assembly which represent the reactor coolant pressure boundary will be monitored for the initiation and progression of sleeve/tube wall degradation, thus satisfying the requirement of Regulatory Guide 1.83. The portion of the tube bridged by the sleeve joints is effectively removed from the pressure boundary, and the sleeve then forms the new pressure boundary. The sleeve enhances the safety of the plant by increasing the protective boundaries of the steam generator. Keeping the tube in service with the use of a sleeve, instead of plugging the tube and removing it from service, increases the heat transfer efficiency of the steam generator. Monitoring for any increased degradation of a repaired steam generator tube shall be implemented by sampling twenty (20) percent of the sleeves inservice. During each scheduled inservice inspection, any sampled sleeve evaluated and found to have unacceptable degradation shall be removed from service.

Based on the preceding analysis it is concluded that operation of Indian Point Unit No. 2 in accordance with the proposed amendment does not increase the probability of an accident previously evaluated, does not create the possibility of a new or different kind of accident from any accident previously evaluated, nor reduce any margin of plant safety. Therefore, the license amendment does not involve a Significant Hazards Consideration as defined in 10 CFR 50.92.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610

Attorney for licensee: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003

NRC Project Director: Ledyard B. Marsh

Duke Power Company, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: September 30, 1994, as supplemented by letter dated September 19, 1995

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) related to the replacement of the steam generators at McGuire, Units 1 and 2. Currently, the steam generators in place at the McGuire units are Westinghouse Model ''D'' type preheat steam generators. The tube degradation levels in the generators has affected the reliability of the units. Therefore, these generators are scheduled to be replaced with feedring steam generators designed by Babcock and Wilcox International.

In the licensee's September 19, 1995, supplement, proposed changes were made to TS Table 2.2-1, ''Reactor Trip System Instrumentation Trip Setpoints,'' to change the programmed T_{AVG} from 588.2 °F to 585.1 °F. This temperature was chosen based on returning the secondary side steam pressure to the original value after replacement of the steam generators. The licensee stated that 585.1 °F was the assumed value for nominal full power T_{AVG} in all applicable safety analyses related to replacement of the steam generators.

The licensee also requested that the steam line safety valve lift settings in Table 3.7-3, which was requested in the September 30, 1994, application, be withdrawn. The licensee determined that these changes are no longer needed.

Basis for proposed no significant hazards consideration determination: As required by 10CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Operation of McGuire Nuclear Station in accordance with the proposed changes to the Technical Specifications will not involve a significant increase in the probability or consequences of an accident previously evaluated. The low-low steam generator water level reactor trip setpoint, the high-high steam generator water level setpoint for turbine trip and feedwater isolation, and the low-low steam generator water level setpoint for auxiliary feedwater initiation are changing to support operation with the replacement steam generators. These setpoints were chosen both to optimize plant operation, and ensure that all applicable acceptance criteria are met for licensing basis safety analysis. These setpoints do not contribute to the initiation of any accident evaluated in the McGuire FSAR [Final Safety Analysis Report] and have no adverse impact on system operation, therefore it can be concluded that these changes will not significantly increase the probability or consequences of an accident evaluated in the FSAR.

The reduction in the primary to secondary leakage rate for McGuire will not increase the probability of an accident evaluated in the FSAR. This lower limit will require corrective action more quickly than is currently required in the event that there is a steam generator tube leak. This change will not significantly affect the consequences of an accident previously evaluated. The allowable leakage is being lowered because this leakage has a major impact on the results of the offsite dose calculation for the locked rotor, single uncontrolled rod withdrawal, and rod ejection events. The taller tube bundle in the replacement steam generators will potentially result in a longer period of tube bundle uncovery during the above transients. The revised allowable leakages of 0.27 gpm through all steam generators and 135 gallons per day through any one generator ensure that the dose analysis results are within the applicable fraction 10 CFR 100 limits.

The increase in Reactor Coolant System volume due to the replacement steam generators will not increase the probability or consequences of an accident previously evaluated. The increase in volume has no effect on the probability of occurrence of any accident evaluated in the FSAR. The mass and energy release due to postulated loss of coolant accidents inside containment has been analyzed to ensure that the peak containment pressure limit is not exceeded. All Chapter 15 reanalysis which was required due to the replacement steam generators assumed the new Reactor Coolant System volume. Since the results of these analyses show the applicable acceptance criteria continue to be met, it can be concluded that the consequences of an accident previously evaluated are not significantly increased due to this change.

* * * *

Operation of McGuire Nuclear Station in accordance with the proposed changes to the Technical Specification will not create the possibility of a new or different accident from any accident previously evaluated. The proposed changes to revise the low-low steam generator water level reactor trip setpoint, high-high steam generator water level setpoint for turbine trip and feedwater isolation, and low-low steam generator water level setpoint for auxiliary feedwater initiation ensure that the appropriate acceptance criteria for FSAR Chapter 15 transients which rely on these functions are met for operation with the replacement steam generators. The proposed change to lower primary to secondary leakage for operation with the replacement steam generators will require that corrective action be taken more quickly in the event that steam generator tube leakage is experienced during operation. As discussed in the technical justification, this will cause the dose results for transients affected by tube bundle uncovery to be within acceptable limits. .... The increase in Reactor Coolant System volume is taken into account in the analysis of the mass and energy release due to a postulated loss of coolant inside containment and Chapter 15 events which have been reanalyzed due to replacement of the steam generators. As discussed above, the proposed changes will not introduce the possibility of a new or different accident from any previously evaluated; they will ensure that transients that take credit for these functions and dose analyses meet applicable acceptance criteria for operation with the replacement steam generators.

Operation of McGuire Nuclear Station in accordance with the proposed changes to the Technical Specifications will not involve a significant reduction in a margin of safety. The proposed changes are being made to ensure that transients that rely on low-low steam generator water level reactor trip setpoint, high-high steam generator water level setpoint for turbine trip and feedwater isolation, and low-low steam generator water level setpoint for auxiliary feedwater actuation meet applicable acceptance criteria. The reduction in allowable primary to secondary leak rate will ensure that transients with dose analyses which are affected by the replacement steam generators meet the current acceptable limits. .... The proposed change in the Reactor Coolant System volume will not involve a significant reduction in a margin of safety. The increased volume affects the mass and energy release due to a postulated loss of coolant accident inside containment and the other Chapter 15 events which were reanalyzed due to replacement of the steam generators. These events have been analyzed and the results are within current acceptable limits. As discussed above, the acceptance criteria for FSAR transients which are affected by these proposed changes continue to be met, therefore there is no significant reduction in the margin of safety.

Changes to the steam generator surveillance requirements will simply delete inspection requirements and repair methods which are no longer applicable after installation of the replacement steam generators. The only exception to this is Surveillance Requirement 4.4.5.4.a.9. This requirement is modified to clarify that the manufacturer will perform the hydrostatic test for the replacement steam generators. This change will not affect the probability or consequences of an accident previously evaluated, the purpose of the preservice inspection is to establish the baseline condition of the tubing. The baseline condition of the tubing in the replacement steam generators will be established prior to installation. The possibility of a new or different accident from any previously evaluated will not be created. No new accident initiation mechanisms will be introduced by this change, and the intent of the requirement, to establish the baseline condition of the tubing, will be met. Since the baseline condition of the tubing will be obtained for use in the monitoring of tubing degradation, as is currently required by the surveillance requirement, there will not be a significant reduction in the margin of safety.

The changes to Technical Specification 6.9.1.9 are administrative in nature. These changes are being made to reflect the most recent revisions of DPC-NE-3002 and DPC-NE-3000, which include changes associated with the replacement steam generators. These topical reports revisions will be reviewed and approved for use regarding Catawba and McGuire Nuclear Stations. Since these changes are administrative in nature, no significant hazards considerations are involved.

Proposed revision to TS Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints:

proposed change to the Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously evaluated. Changing the value for T_{AVG} in Notes 1 and 2 of Table 2.2-1 will update the value to agree with the T_{AVG} assumed in the applicable safety analyses for replacement of the steam generators. Acceptable results were obtained for all required reanalyses. The probability of an accident will not be significantly affected by operation with the new T_{AVG} value, because all equipment will be operated within acceptable design limits. The consequences of previously evaluated accidents which are affected by this change have been evaluated, and have been determined to be within acceptable limits.

This proposed change will not create the possibility of a new or different kind of accident from any previously evaluated. This change does not change the physical configuration of the plant, and all analyses which are affected by replacement of the steam generators have been determined to have acceptable results assuming this value for T_{AVG}.

This proposed change to the Technical Specifications will not involve a significant reduction in the margin of safety. All safety analyses which were affected by replacement of the steam generators assumed this value for T_{AVG} and the results were determined to be within previously acceptable limits.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242

NRC Project Director: Herbert N. Berkow

Entergy Operations, Inc., Docket Nos. 50-313 and 50-368, Arkansas Nuclear One, Unit Nos. 1 and 2 (ANO-1&2), Pope County, Arkansas

Date of amendment request: September 4, 1993, as supplemented on February 16, 1994, and August 4, 1995.

Description of amendment request: The proposed amendment would revise the Arkansas Nuclear One Industrial Security Plan.

Basis for proposed no significant hazards consideration determination: As required by 10CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below.

The accident mitigation features of the plant are not affected by the proposed compensatory measures for protecting the site during periods when security systems are degraded and therefore no decrease occurs in the effectiveness of the security program to protect against radiological sabotage or increased risk to the public health and safety. This is due to continued compliance with existing regulatory requirements and other commitments within the security plan. These changes have no impact on the design basis security threat and accordingly do not create the possibility of a new or different kind of accident. New systems, modes of equipment operation, failure modes or other plan situations are not introduced by these changes. The proposed changes allow flexibility for the use of compensatory measures and do not change any safety limits, LCOs, or surveillance requirements on equipment to operate the plant.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005-3502

NRC Project Director: William D. Beckner

Gulf States Utilities Company, Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: October 24, 1995, as supplemented or supercedes letters dated May 30, and June 20, 1995

Description of amendment request: The proposed amendment would revise the technical specifications (TSs) on containment systems to reflect the adoption of requirements of 10 CFR Part 50, Appendix J, Option B, and implementation of a performance-based containment leak rate testing program at River Bend Station. The licensee letters dated May 20, and June 20, 1995, requested an exemption to Appendix J which subsequently became Option B to the appendix. Those letters were noticed in the Federal Register on July 5, 1995 (60 FR 35079).

Basis for proposed no significant hazards consideration determination: As required by 10CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. This request does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that contribute to initiation of any accidents previously evaluated. Thus, the proposed change cannot increase the probability of any accident previously evaluated.

The proposed change potentially affects the leak-tight integrity of the containment structure designed to mitigate the consequences of a loss-of coolant accident (LOCA). The function of the containment is to maintain functional integrity during and following the peak transient pressures and temperatures which result from any loss-of-coolant accident (LOCA)[LOCA]. The containment is designed to limit fission product leakage following the design basis LOCA. Because the proposed change does not alter the plant design, only the frequency of measuring Type B and C leakage, the proposed change does not directly result in an increase in containment leakage. However, decreasing the test frequency can increase the probability that a large increase in containment leakage could go undetected for an extended period of time. Based upon the results of the periodic containment Type A or Integrated Leak Rate Tests (ILRTs) and Type B and C or Local Leak Rate Tests (LLRTs) surveillance tests, this is not expected during the remaining life of the plant. The risk resulting from the proposed changes is as follows:

Type A Testing

NUREG/CR-4330 (NRC86) found that the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of the containment. It also determined that on an expected individual dose basis, the effect of containment leakage is small.

Industry wide, ILRTs have only found a small fraction of the leaks that exceed current acceptance criteria. Only three percent of all leaks have a potential for remaining undetected for longer periods of time. In addition, when leakage has been detected by ILRTs, the leakage rate has been only about two times the allowable leakage rate.

NUREG-1493 found that these observations, together with the insensitivity of reactor accident risk to the containment leakage rate, show that reducing the Type A leakage test frequency would have a minimal impact on public risk.

Type B and C Testing

NUREG-1493 found that while Type B and C tests can identify the vast majority (greater than 95 percent) of all potential leakage paths, performance-based alternatives to current local leakage-testing requirements are feasible without significant risk impacts. The risk model used in NUREG-1493 suggests hat the number of components tested would be reduced by about 60 percent with less than a three-fold increase in the incremental risk due to containment leakage. Since, under existing requirements, leakage contributes less than 0.1 percent of overall accident risk, the overall impact is very small. NUREG-1493 found that while the extended testing intervals for Type B and C tests led to minor increases in potential offsite [off-site] dose consequences, the actual increase in on-site (worker) doses exceeded (by at least an order of magnitude) the potential off-site dose consequences.

EPRI Research Project Report TR-104285, ''Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals,'' also concluded that a relaxation of the test intervals for Type B and C penetrations results in a negligible increase in total plant risk.

Based on the above EOI [Entergy Operation, Inc.] has concluded that the proposed change will not result in a significant increase in the probability or consequences of any accident previously evaluated.

2. The request does not create the possibility of occurrence of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to initiation of any accidents. This change involves the reduction in Type B and C test frequency. The methods of performing the tests are not changed. No new accident modes are created by extending the testing intervals. No safety-related equipment or safety functions are altered as a result of this change. Extending the test frequency has no influence on , nor does it contribute to, the possibility of a new or different kind of accident or malfunction from those previously analyzed.

3. The request does not involve a significant reduction in a margin to safety.

The proposed change only affects the frequency of Type A, B, and C testing and does not change the methodology for performance of the testing. However, the proposed change can increase the probability that a large increase in leakage could go undetected for an extended period of time. Operational experience has shown that the leak tightness of the containment has been maintained significantly below the allowable leakage limit. In addition, NUREG-1493 has determined that, under several different accident scenarios, the risk of radioactivity release from containment is negligible with the implementation of these proposed changes.

The margin of safety that has the potential of being impacted by the proposed change involves the offsite [off-site] dose consequences of postulated accidents which are directly related to containment leakage rate. The containment isolation system is designed to limit leakage to L_a which is defined by the RBS Technical Specifications to be 0.26 percent by weight of the containment air per 24 hours at 7.6 psig (P_a). The limitation on containment leakage rate is designed to ensure that total leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure (P_a) or 7.6 psig. The margin to safety for the offsite [off-site] dose consequences of postulated accidents directly related to the containment leakage rate in maintained by meeting the 1.0 L_a

No change in the method of testing is being proposed. The Type B and C tests will continue to be done at full pressure (P_a) or greater. Other programs are in place to ensure that proper maintenance and repairs are performed during the service life of the primary containment and systems and components penetrating the primary containment.

As a result, EOI had concluded that the proposed change will not result in a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005

NRC Project Director: William D. Beckner

Public Service Electric & Gas Company, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: September 28, 1995

Description of amendment request: The proposed change modifies Technical Specification 3/4.8.1.2, ''Electrical Power Sources - Shutdown.'' The surveillance requirement 4.8.1.2 is clarified by a Note to identify those surveillances which are required to be performed during Modes 5 and 6.

Basis for proposed no significant hazards consideration determination: As required by 10CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Will not involve a significant increase in the probability or consequences of an accident previously evaluated.

No component modification, system realignment, or change in operations will occur which could affect the probability of any accident or transient. The proposed addition of a Note will provide guidance on which surveillances are required to be performed in Modes 5 and 6. The Note will preclude rendering operable DGs inoperable, and/or preclude de-energizing a required ESF bus or disconnecting a required offsite circuit during the performance of the surveillance requirement. Proposed changes do not eliminate any testing requirements, they simply clarify which tests will be performed in Modes 5 and 6, and which are required to be performed prior to entry into Mode 4. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Will not create the possibility of a new or different kind of accident from any previously evaluated.

No component modification, system realignment, or change in operating procedure is required to implement the proposed change. The proposed change reduces the possibility of a single event impacting the operability of an ESF bus or its DG simultaneously. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will not involve a significant reduction in a margin of safety.

The proposed change will not alter any assumptions, initial conditions, or results of any accident analyses. The Class 1E equipment assumed available in the accident analyses and their designed capability to mitigate the consequences of any postulated accidents will not be changed. The addition of a Note to clarify the surveillance requirements will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Salem Free Public library, 112 West Broadway, Salem, New Jersey 08079

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston and Strawn, 1400 L Street, NW, Washington, DC 20005-3502

NRC Project Director: John F. Stolz

Public Service Electric & Gas Company, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: September 28, 1995

Description of amendment request: The proposed changes relocate ''Reactor Coolant System - Chemistry'' Technical Specification 3/4.4.7 (Salem Unit 1) and 3/4.4.8 (Salem Unit 2) and their associated Bases to the Salem Updated Final Safety Analysis Report (UFSAR) and the Surveillance Requirements and Limiting Conditions for Operation to applicable plant procedures controlled by the 10 CFR 50.59 process. Also, the applicability will be changed from ''At all times'' to ''Modes 1, 2, 3, 4, 5 and 6.''

Basis for proposed no significant hazards consideration determination: As required by 10CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes involve no hardware changes, no changes to the operation of any systems or components, and no changes to existing structures. Specifically, changing the Applicability from ''At all times'' to ''Modes 1, 2, 3, 4, 5 and 6'' by this submittal will not alter established chemistry for chlorides, fluorides and dissolved oxygen of the Reactor Coolant System. The relocation of this Surveillance Requirement/LCOs and Bases to plant procedures and the UFSAR respectively, will continue to ensure that the chemistry analysis of the Reactor Coolant System water is monitored and controlled. Changing the Applicability from ''At all times'' to ''Modes 1,2,3,4,5 and 6'' represent changes that do not affect plant safety and do not alter existing accident analyses.

2. Will not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes are procedural in nature concerning the location of the descriptive information and surveillance requirements for Reactor Coolant System Chemistry. Removing these specifications from the Technical Specifications and placing them in the UFSAR and plant procedures will not alter the maintenance of the Reactor Coolant System Chemistry or the ability to monitor its intended functions. Therefore, these changes will not create a new or unevaluated accident or operating condition.

3. Will not involve a significant reduction in a margin of safety.

The proposed changes relocate the Reactor Coolant System Chemistry Requirements/LCOs from the Technical Specifications to the UFSAR and plant procedures in accordance with guidance provided by the NRC Final Policy Statement (58 FR 39132) regarding the improvement of Technical Specifications. The requirements that will reside in the UFSAR and plant procedures for the Reactor Coolant System Chemistry will ensure that the ability to determine chloride, fluoride and dissolved oxygen concentrations in the Reactor Coolant System is properly maintained and that the maintenance of the Reactor Coolant System Chemistry will be commensurate with its safety significance. Therefore, the proposed changes will not involve a significant reduction in any margins of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Salem Free Public library, 112 West Broadway, Salem, New Jersey 08079

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston and Strawn, 1400 L Street, NW, Washington, DC 20005-3502

NRC Project Director: John F. Stolz

Southern Nuclear Operating Company, Inc., Docket Nos. 50-348 and 50-364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendments request: September 26, 1995

Description of amendments request: The amendments would revise Technical Specification (TS) Section 4.6.1.3 to incorporate improvements to containment air lock testing referenced in Chapter 3.6, ''Containment Systems,'' of NUREG-1431, ''Standard Technical Specifications, WestinghousePlants.''

Basis for proposed no significant hazards consideration determination: As required by 10CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change does not involve any change to the configuration or method of operation of any plant equipment used to mitigate the consequences of an accident. Containment leakage is an assumption in the safety analysis of the loss of coolant accident and the rod ejection accident. Changes to the containment air lock door seal test acceptance criteria will have no impact on the radiological consequences of these accidents since the plant safety analysis is based on the assumption that the containment leaks at its design leak rate of 0.15 percent per day for the first 24 hours and 0.075 percent per day thereafter for each of these accidents. The change to the surveillance requirement meets the intent of the guidance in NUREG-1431. Primary containment integrity ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analysis. The limitations on closure and leak rate for the containment air locks are required to meet these restrictions on containment integrity. These changes do not increase the probability that the 10 CFR [Part] 100 limits will be exceeded. The change to the surveillance requirement does not impose any new safety analyses limits or alter the plants ability to detect and mitigate events. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change involves a revision to the Technical Specifications to meet the intent of the guidance of NUREG-1431, and does not necessitate a physical alteration of the plant or change in parameters governing normal plant operation. The change has not effect on the plant's compliance with the requirements of Appendix J. The revision of the acceptance criteria for the air lock door seal test will improve the FNP [Farley Nuclear Plant] current testing criteria while maintaining an acceptable level of safety. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety. The revision of the acceptance criteria of the air lock door seal test will decrease the overall test burden without decreasing the margin of safety. The overall leakage rate of the air lock continues as less than or equal to $0.05L_a$ and the plant safety analysis continues to be based ont he assumption that the containment leaks at its design leak rate of 0.15 percent per day for the first 24 hours and 0.075 percent per day thereafter for each of these accidents. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201

NRC Project Director: Herbert N. Berkow

Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company, Docket No. 50-346, Davis-Besse Nuclear Power Station, Unit No. 1, Ottawa County, Ohio

Date of amendment request: July 28, 1995

Description of amendment request: The proposed amendment would clarify the limiting condition for operation for TS 3.8.1.1 and 3.8.1.2 from ''independent'' circuit to ''qualified'' circuit; explain in the Bases the requirements for operability of an offsite circuit; delete the STAGGERED TEST BASIS scheduling requirement to perform emergency diesel generatorsurveillances; explain in the Bases an acceptable method for verification of Emergency Diesel Generator speed for surveillance requirements (SR) 4.8.1.1.2.a.4 and 4.8.1.1.2.c.4; remove a surveillance test extension that has expired for SR 4.8.1.1.1.b; add an exception for SR 4.8.1.1.2.c.5 and 4.8.1.1.2.c.7 to SR 4.8.1.2; and revise Bases 3.0.5 to reflect the clarification from ''independent'' circuit to ''qualified'' circuit.

Basis for proposed no significant hazards consideration determination: As required by 10CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Toledo Edison has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation ofthe Davis-Besse Nuclear Power Station, Unit No. 1 in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because the proposed changes do not make a change to any accident initiator, initiating condition or assumption. The proposed changes do not involve a significant change to the plant design or operation. The proposed changes do not affect the safety function of the offsite circuits or the emergency diesel generators (EDGs).

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not invalidate assumptions used in evaluating the radiological consequences of an accident, do not alter the source term or containment isolation and do not provide a new radiation release path or alter potential radiological releases.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not introduce a new or different accident initiator or introduce a new or different equipment failure mode or mechanism.

3. Not involve a significant reduction in a margin of safety because the proposed changes do not reduce the margin to safety which exists in the present Technical Specifications [TS] or Updated Safety Analysis Report. The operability requirements of the TS are consistent with the initial condition assumptions of the safety analyses. Further, the proposed changes do not affect the Action statement requirements for the various levels of degradation in the offsite [power] circuits or EDGs.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606

Attorney for licensee: Jay E. Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Project Director: Gail H. Marcus

Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company, Docket No. 50-346, Davis-Besse Nuclear Power Station, Unit No. 1, Ottawa County, Ohio

Date of amendment request: September 29, 1995

Description of amendment request: The proposed amendment would increase the minimum available borated water volume requirement for the boric acid addition system, the minimum and maximum boron concentration requirements for the borated water storage tank, the minimum boron concentration requirement for the core flood tanks; modify the surveillance requirements for trisodium phosphate dodecahydrate; and modify the refueling boron concentration and the associated Action statement. These proposed changes will affect the following Technical Specification sections: 3/4.1.2.8, Reactivity Control Systems - Borated Water Sources - Shutdown; 3/4.1.2.9, Reactivity Control Systems - Operating; 3/4.5.1, Emergency Core Cooling Systems (ECCS) - Core Flooding Tanks; 3/4.5.2, Emergency Core Cooling Systems - ECCS Subsystems - Tavg [plus or minus] 280 °F; 3/4.5.4, ECCS - Borated Water Storage Tank; 3/4.9.1, Refueling Operations - Boron Concentration; Bases 3/4.1.2, Boration Systems; Bases 3/4.5.2 and 3/4.5.3, ECCS Subsystems; and Bases 3/4.9.1 Boron Concentration.

Basis for proposed no significant hazards consideration determination: As required by 10CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Toledo Edison has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no accident initiators, conditions, or assumptions are significantly affected by the proposed changes.

The proposed changes to the Technical Specifications and their Bases increase the minimum volume of the Boric Acid Addition System (BAAS), the minimum boron concentration of the Borated Water Storage Tank (BWST) and Core Flooding Tanks (CFTs), the maximum boron concentration of the BWST, and the minimum volume of trisodium phosphate dodecahydrate (TSP) in Containment (CTMT). Administrative changes to these Technical Specifications have also been proposed. These changes ensure adequate boration capability is maintained for normal operations, that adequate Shutdown Margin (SDM) can be achieved following an accident, and that the assumed post-Loss of Coolant Accident (LOCA) pH can be achieved. Therefore, as stated above, these proposed changes do not significantly affect accident initiators, conditions, or assumptions.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not change the source term, CTMT isolation, or allowable releases.

In particular, maintaining the appropriate amount of TSP will ensure the assumed pH will be achieved, the assumption of source term with respect to iodine retention will be maintained, and the radiological consequences of a previously evaluated accident will not be increased.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes.

These changes ensure that the assumptions used for initial and final conditions of SDM, pH, and source term are maintained. Also, the Environmental Qualification (EQ) and seismic requirements have been verified to be adequate to maintain the adequacy of Structures, Systems, and Components (SSCs) during assumed accident conditions.

3. Not involve a significant reduction in a margin of safety because the proposed changes to the minimum volume and boron concentration for the BAAS, BWST, and CFTs ensure that the margin of safety for reactor subcriticality is maintained at all times for future longer fuel cycles, including the upcoming Cycle 11.

The proposed increase in the BWST maximum boron concentration is set at the conservative limit for post-LOCA boron precipitation concerns. Therefore, the existing margin of safety with respect to post-LOCA boron precipitation is maintained.

The proposed increase in the minimum TSP volume requirement maintains the same margin of safety with respect to post-LOCA pH, time for dissolution, iodine retention, and chloride stress corrosion of austenitic stainless steels. The TSP capacity margin of approximately 40 cubic feet included in the minimum TSP volume requirement will not result in increasing the pH above the previously approved pH limit of 11. This reserve capacity adds margin to ensure adequate minimum pH is achieved.

The proposed removal of the 1800 ppm refueling boron concentration requirement does not reduce the margin of safety because the requirement of maintaining keff [less than or equal to] 0.95 is alone sufficient to ensure that the accident analysis assumptions are satisfied.

The proposed change to the boration rate requirement of the

LCO 3.9.1 Action statement does not reduce the margin of safety because the proposed boration rate of 12 gpm of 7875 ppm boric acid solution is equivalent to the present boration rate of

10 gpm of 8750 ppm boric acid solution.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606

Attorney for licensee: Jay E. Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Project Director: Gail H. Marcus

Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company, Docket No. 50-346, Davis-Besse Nuclear Power Station, Unit No. 1, Ottawa County, Ohio

Date of amendment request: October 2, 1995

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 5.0, ''Design Features,'' by adding a site location description, remove site area maps, remove containment and reactor coolant system design parameters, remove the description of the meteorological tower location, remove component cyclic or transient limits, and revise the fuel assembly description to include the use of ZIRLO clad fuel rods.

Basis for proposed no significant hazards consideration determination: As required by 10CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Toledo Edison has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station Unit Number 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no accident initiators, conditions or assumptions are affected by the proposed changes to Section 5.0, Design Features, of the Technical Specifications. These changes are proposed to add a site location description, remove site area maps, remove containment and reactor coolant system design parameters, remove the description of the meteorological tower location, remove component cyclic or transient limits, and revise the fuel assembly description to include the use of ZIRLO clad fuel rods.

Under the proposed changes, Technical Specifications (TS) Section 5.0 would continue to satisfy the applicable requirements of Section 182.a of the Atomic energy Act of 1954, and 10 CFR 50.36(c)(4). Further, the proposed changes are consistent with NUREG-1430, ''Standard Technical Specifications for Babcock and Wilcox Plants,'' Revision 1. The information proposed for removal from existing TS 5.0 is presently included in the Updated Safety Analysis Report (USAR) or is being proposed to be added to the USAR, hence sufficient controls exist under 10 CFR 50.59 to ensure that future changes to these items are acceptable.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because no accident conditions or assumptions are affected by the proposed changes. As described above, these changes are consistent with the ''Standard Technical specifications for Babcock and Wilcox Plants'' (NUREG-1430) and are administrative changes. The proposed changes do not alter the source term, containment isolation, or allowable releases. The proposed changes, therefore, will not increase the radiological consequences of a previously evaluated accident.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes, which involve only administrative controls. As described above, these changes are consistent with the ''Standard Technical Specifications for Babcock and Wilcox Plants'' (NUREG-1430) and are administrative changes. The proposed changes do not alter any accident scenarios.

3. Not involve a significant reduction in a margin of safety because the proposed changes are administrative and do not reduce or adversely affect the capabilities of any plant structure, systems or components.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606

Attorney for licensee: Jay E. Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Project Director: Gail H. Marcus

Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of amendment request: September 6, 1995

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 5.3.1 to reflect a change in the maximum initial enrichment for reload fuel. The amendment would also change the maximum reference $K_{infinity}$ for storage in Region 1 of the spent fuel pool and TS Figure 3.9-1 to reflect a change in the maximum initial enrichment for storage in Region 2.

Basis for proposed no significant hazards consideration determination: As required by 10CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

An increase to a maximum initial enrichment of 5.0 w/o U-235 does not involve an increase in the probability or consequence of an accident or other adverse condition over previous evaluations. Because of the conservative techniques and assumptions used to evaluate the maximum possible neutron multiplication factor, there is reasonable assurance that criticality safety is maintained when storing fuel assemblies of up to and including 5.0 w/o U-235 in the spent fuel storage racks under both normal and postulated accident conditions. For example, the calculations for non-accident conditions ignore the 2000 ppm soluble boron in the spent fuel pool calculations, thus resulting in conservative values of the multiplication factor. Storing fuel in the Region 1 configuration which meets the IFBA [integral fuel burnable absorber] versus enrichment curve (Figure 3 of Attachment 6) results in a maximum multiplication factor of 0.9481, including all biases and uncertainties.

2. The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

An increase to a maximum initial enrichment level of 5.0 w/o U-235 does not create the possibility of a new or different kind of accident or condition over previous evaluations. An increase to the enrichment level of 5.0 w/o U-235 involved performing extensive evaluations to develop the IFBA versus enrichment curve for V-5 fuel. Use of dual code packages ensures that the spent fuel pool Region 1 criticality limits are not exceeded.

3. The proposed changes do not involve a significant reduction in the margin of safety.

An increase in the maximum initial enrichment level to 5.0 w/o U-235 does not involve a reduction in the margin of safety. As discussed above, in all cases the multiplication factors for worst case assumptions fall considerably below the criticality limits and do not represent any reductions in margin. An increase to the initial enrichment level of 5.0 w/o U-235 does not adversely impact operation of the various plant systems, i.e. HVAC [heating, ventilation, and air conditioning], spent fuel pool cooling, or radiological control systems.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Project Director: William H. Bateman

Wisconsin Public Service Corporation, Docket No. 50-305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of amendment request: October 6, 1995

Description of amendment request: The proposed amendment would revise Kewaunee Nuclear Power Plant (KNPP) Technical Specification (TS) 4.2.b, ''Steam Generator Tubes,'' its associated bases, and Figure TS 4.2-1 by redefining the pressure boundary for Westinghouse mechanical hybrid expansion joint (HEJ) steam generator (SG) tube sleeves.

Basis for proposed no significant hazards consideration determination: As required by 10CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed change was reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist.

1. Operation of the KNPP in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Mechanical testing has shown that the inherent structural strength of the HEJ joint provides sufficient integrity such that the tube rupture capability recommendations of RG [Regulatory Guide] 1.121 are met, even for instances of 100 percent throughwall, 360° circumferentially oriented degradation in the HEJ HRLT [hardroll lower transition] region. Structural integrity recommendations consistent with RG 1.121 are supplied

for all tube degradation 1.1 inch or greater below the bottom of the HEJ HRUT [hardroll upper transition]. Based on test data, a bounding SLB [steam line break] leak rate of 0.033 gpm for indications between 1.1 and 1.3 inch below the bottom of the HRUT is applied. As the leakage data base is expanded and statistical basis established, this SLB leakage allowance may be reduced. For indications existing greater than 1.3 inch below the bottom of the HRUT, SLB event leakage can be neglected.

Additional prevention from tube rupture is inherently provided by the HEJ geometry. For RCS [reactor coolant system] release rates to exceed the normal makeup capacity of the plant, the tube must be postulated to experience a complete circumferential separation at the lower transition, and become axially displaced by 3 to 3.25 inches, resulting in complete geometric disassociation between the tube and sleeve resulting in sufficient flow area to support leakage in excess of makeup capacity. During the 1989 plug top release event at North Anna Unit 1, primary to secondary release rates were calculated to be less than 80 gpm, for a flow area approximately four times larger than the flow area created by a tube which was axially displaced by about 1.25 to 1.5 inch. Analysis of the steam generator indicates that at a 95 percent cumulative probability, the tube would experience an axial displacement of less than the 1.1 inch boundary. At this level of axial displacement, a ring of metal to metal contact would remain between the tube and sleeve, and leakage would be far less than makeup. Projected leakage at this point is expected to be less than 2.5 gpm. Therefore, implementation of the proposed repair boundary will not result in tube rupture, even for a tube postulated to not behave as predicted by the available test and pulled tube data.

The proposed technical specification change to support the implementation of the HEJ sleeve tube pressure boundary for parent tube degradation in the HEJ HRLT region does not adversely impact any other previously evaluated design basis accident or the results of accident analyses for the current technical specification minimum reactor coolant system flow rate. Plugging limit criteria are established using the guidance of RG 1.121. Furthermore, per RG 1.83 recommendations, the sleeved tube assembly can be monitored through periodic inspections with present eddy current techniques.

2. The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Implementation of the revised pressure boundary will not introduce significant or adverse changes to the plant design basis. Mechanical testing of degraded sleeve joints supports the conclusions of the calculations that the sleeve retains structural (tube burst) capability consistent with RG 1.121. As with initial installation of sleeves, implementation of the relocated pressure boundary cannot interact with other portions of the RCS. Any hypothetical accident as a result of potential tube degradation in the HEJ HRLT region of the tube is bounded by the existing tube rupture accident analysis. Neither the sleeve design nor implementation of the tube repair boundary defined on Figure TS 4.2-1 affects any other component or location of the tube outside of the immediate area repaired.

3. The proposed license amendment does not involve a significant reduction in a margin of safety.

The safety factors used in the establishment of the HEJ sleeved tube pressure boundary are consistent with the safety factors in the ASME [American Society of Mechanical Engineers] Boiler and Pressure Vessel Code used in steam generator design. Based on the sleeved tube geometry, it is unrealistic to consider that application of the revised pressure boundary could result in single tube leak rates exceeding the normal makeup capacity during normal operating conditions. The pressure boundary established ... has been developed using the methodology of RG 1.121. The performance characteristics of postulated degraded parent tubes of HEJ tube/sleeve joints have been verified by testing to retain structural integrity and preclude significant leakage during normal and postulated accident conditions. Testing indicates that postulated circumferentially separated tubes which the repair boundary addresses would not experience axial displacement during either normal operation or SLB conditions. The existing offsite dose evaluation performed for KNPP in support of the voltage based plugging criteria for axial ODSCC [outside diameter stress corrosion cracking] at TSP [tube support plate] intersections established a faulted loop primary to secondary leak rate of 34.0 gpm using technical specification dose equivalent Iodine-131 activity levels. Following implementation of the criteria, postulated leakage from all sources must not exceed 34.0 gpm in the faulted loop. Maintenance of this limit will ensure that offsite doses would not exceed the currently accepted limit of a small fraction of the 10 CFR 100 guidelines. The repair boundary uses a conservatively established ‘‘per indication’’ leak rate for estimation of SLB leakage. This leak rate is applied to all indications left in service as a result of the tube repair boundary, including non-throughwall indications and a limited number of indications of circumferential throughwall extent.

For a postulated indication whose performance is not characteristic of the test and pulled tube data, and which would experience axial displacement at the 95 percent cumulative probability value following a postulated SLB event with no operator intervention, leakage would not be expected to result in an uncontrolled release of reactor coolant in excess of normal makeup capacity.

For the three removed tube sleeve samples and nearly 1,000 PTIs [parent tube indications] detected in the field, there were no instances of degradation of elevations (multiple expansion transitions) on either side of the hardroll expansion in the same tube. This includes no instances on non-detected degradation in the upper hydraulic and hardroll upper expansion transitions for the removed tubes. One tube was identified in the most recent KNPP inspection with two separate circumferential crack elevations within the HRLT. Rapidly occurring degradation would not be expected at the upper transitions, based partly on the field inspection results. The available inspection results include two inspection programs (1994 and 1995) at Kewaunee and one at Point Beach Unit 2 (1994). Through these three inspection programs, approximately 11,000 HEJ sleeved tubes have been inspected using advanced ET [eddy current testing] techniques.

The portions of the installed sleeve assembly which represent the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation, thus satisfying the requirements of Regulatory Guide 1.83.

The NRC staff has reviewed the licensee’s analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, Wisconsin 54311-7001

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P. O. Box 1497, Madison, Wisconsin 53701-1497

NRC Project Director: Gail H. Marcus

Previously Published Notices Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the Federal Register on the day and page cited. This notice does not extend the notice period of the original notice.

Commonwealth Edison Company, Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station, Units 1 and 2, Lake County, Illinois

Date of amendment request: October 6, 1995

Description of amendment request: Revise the Technical Specifications to change the definition of the F* distance.

Date of publication of individual notice in Federal Register: October 16, 1995 (60 FR 53648)

Expiration date of individual notice notice: November 15, 1995

Local Public Document Room location: Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085

Wisconsin Electric Power Company, Docket Nos. 50-266 and 50-301, Point Beach Nuclear Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of application for amendment: September 13, 1995, as supplemented by letter dated October 19, 1995

Brief description of amendment request: The proposed amendments would revise Technical Specification (TS) Section 15.1, ''Definitions,'' the basis for TS Section 15.3.1.G, ''Operational Limitations,'' and TS Figure 15.2.1-2, ''Reactor Core Safety Limits, Point Beach Unit 2.'' The proposed changes would reduce the reactor coolant system raw measured total flow rate limit and reflect new reactor core safety limits for Unit 2.Date of individual notice in Federal Register: October 24, 1995 (60 FR 54527)

Expiration date of individual notice notice: November 8, 1995

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin

Notice Of Issuance Of Amendments To Facility Operating Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the Federal Register as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: June 6, 1995

Brief description of amendments: The amendments extend the nominal surveillance interval requirements of selected safety systems instruments form 18 months to a refueling interval of 24 months.

Date of issuance: October 19, 1995

Effective date: As of the date of issuance to be implemented within 30 days for Unit 2 and prior to restart of the spring 1996 refueling outage for Unit 1.

Amendment Nos.: 208 and 186

Facility Operating License Nos. DPR-53 and DPR-69: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35061) The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated October 19, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678

Commonwealth Edison Company, Docket Nos. STN 50-454 and STN 50-455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois Docket Nos. STN 50-456 and STN 50-457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois Docket Nos. 50-10, 50-237 and 50-249, Dresden Nuclear Power Station, Units 1, 2 and 3, Grundy County, Illinois Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station, Units 1 and 2, Lake County, Illinois

Date of application for amendments: April 24, 1995, as supplemented August 1 and September 14, 1995.

Brief description of amendments: The amendments would relocate the requirements for the ''Review, Investigative and Audit Functions'' and frequencies of the quality assurance (QA) program from the administrative controls section of the TS to the appropriate sections of the licensee's Quality Assurance Topical Report (QATR), CE-1-A, Revision 65. In addition, the proposed TS changes include title changes to reflect the reorganization of the licensee's Nuclear Operations Division and miscellaneous administrative and editorial changes.

Date of issuance: October 20, 1995

Effective date: October 20, 1995

Amendment Nos.: 75, 75, 67, 67, 38, 141, 135, 107, 93, 163, 159, 171, and 158

Facility Operating License Nos. NPF-37, NPF-66, NPF-72, NPF-77, DPR-2, DPR-19, DPR-25, NPF-11 NPF-18, DPR-29, DPR-30, DPR-39 and DPR-48: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 30, 1995 (60 FR 45175) and September 20, 1995 (60 FR 48726). The August 1 and September 14, 1995, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original Federal Register notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 20, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481; for Dresden, Morris Area Public Library District, 604

Liberty Street, Morris, Illinois 60450; for LaSalle, Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021; and for Zion, Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085

Commonwealth Edison Company, Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station, Units 1 and 2, Lake County, Illinois

Date of application for amendments: June 30, 1995

Brief description of amendments: The amendments modify the surveillance requirements for the emergency diesel generators.

Date of issuance: October 16, 1995

Effective date: October 16, 1995

Amendment Nos.: 170 and 157

Facility Operating License Nos. DPR-39 and DPR-48: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: September 13, 1995 (60 FR 47615) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 16, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085

Entergy Operations, Inc., Docket No. 50-313, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: April 4, 1995

Brief description of amendment: The amendment deletes requirements associated with part length control element assemblies.

Date of issuance: October 12, 1995

Effective date: October 12, 1995

Amendment No.: 169

Facility Operating License No. NPF-6. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 19, 1995 (60 FR 37090) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 12, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of application for amendment: March 17, 1995

Brief description of amendment: The amendment deletes requirements associated with surveillance to verify position stops for High Pressure Safety Injection Emergency Core Cooling System throttle valves.

Date of issuance: October 18, 1995

Effective date: October 18, 1995

Amendment No.: 170

Facility Operating License No. NPF-6. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 19, 1995 (60 FR 37089) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 18, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

Entergy Operations, Inc., Docket No. 50-313, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: April 4, 1995, as supplemented by letter dated October 12, 1995

Brief description of amendment: The amendment revises the containment cooling response time to reduce the likelihood of a water hammer event in service water piping.

Date of issuance: October 26, 1995

Effective date: October 26, 1995

Amendment No.: 171

Facility Operating License No. NPF-6. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 19, 1995 (60 FR 37090) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 26, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

Florida Power and Light Company, et al., Docket Nos. 50-335 and 50-389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of application for amendments: February 28, 1994

Brief description of amendments: The amendments delete the minimum frequency criteria prescribed for quality assurance audits from Administrative Controls sections 6.5.2.8 and 6.8.4 of the Technical Specifications (TS). Audit periodicity will thereby be controlled by the program described in the Florida Power and Light Company (FPL) Topical Quality Assurance Report.

Date of issuance: October 25, 1995

Date of issuance: October 25, 1995

Effective date: October 25, 1995

Amendment Nos.: 140 and 80

Facility Operating License Nos. DPR-67 and NPF-16: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: April 13, 1994 (59 FR 17599) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 25, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

Florida Power and Light Company, Docket Nos. 50-250 and 50-251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of application for amendments: July 26, 1995

Brief description of amendments: These amendments consist of administrative corrections and clarifications.

Date of issuance: October 17, 1995

Effective date: October 17, 1995

Amendment Nos. 177 and 171Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: September 13, 1995 (60 FR 47619) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 17, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199

Florida Power and Light Company, Docket Nos. 50-250 and 50-251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of application for amendments: July 26, 1995

Brief description of amendments: These amendments consist of administrative corrections and clarifications.

Date of issuance: October 17, 1995

Effective date: October 17, 1995

Amendment Nos.: 178 and 172Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: September 13, 1995 (60 FR 47619) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 17, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199

IES Utilities Inc., Docket No. 50-331, Duane Arnold Energy, Center, Linn County, Iowa

Date of application for amendment: February 13, 1995, as supplemented April 21, 1995, and August 7, 1995.

Brief description of amendment: The proposed amendment deletes the audit requirements from the Duane Arnold Energy Center Technical Specifications (TS) and adds them to the Quality Assurance Program.

Date of issuance: October 17, 1995

Effective date: October 17, 1995

Amendment No.: 213

Facility Operating License No. DPR-49. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: March 29, 1995 (60 FR 16190) The additional information contained in the supplemental letters dated April 21, 1995, and August 7, 1995, was clarifying in nature and did not change the NRC staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 17, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, S.E., Cedar Rapids, Iowa 52401

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of application for amendments: March 31, 1995

Brief description of amendments: The amendments revise Technical Specification (TS) surveillance requirements for safety-related pump testing to eliminate recirculation alignments. In addition, specific test parameters, discharge pressures, and flows associated with these pumps are removed from the TS and will be controlled by the Inservice Testing Program.

Date of issuance: October 17, 1995

Effective date: October 17, 1995, with full implementation within 45 days

Amendment Nos.: 203 and 188

Facility Operating License Nos. DPR-58 and DPR-74. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: June 21, 1995 (60 FR 32368) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 17, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085

Niagara Mohawk Power Corporation, Docket No. 50-220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of application for amendment: February 1, 1995

Brief description of amendment: The amendment revises Technical Specification 3.6.13 and associated Bases to permit the controls and instruments from both Remote Shutdown Panels to be considered when assuring that one complete set of controls and instruments is operable. The changes also allow 30 days to restore an inoperable function to operable status, remove MODE 3 (hot shutdown) from the existing requirement for operability, and revise the LIMITING CONDITION FOR OPERATION ACTION to require achieving hot shutdown in 12 hours instead of cold shutdown in 36 hours. An additional change permits the operator 30 days to establish an alternate method of monitoring a parameter (and 90 days to restore the function) when the function is inoperable.

Date of issuance: October 16, 1995

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 155

Facility Operating License No. DPR-63: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: March 1, 1995 (60 FR 11135) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 16, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126

Northeast Nuclear Energy Company, et al., Docket No. 50-423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: March 29, 1995

Brief description of amendment: The amendment modifies the current Technical Specifications that have cycle-specific parameter limits in the Core Operating Limits Report to include an additional cycle-specific parameter and its supporting methodologies.

Date of issuance: October 18, 1995

Effective date: As of the date of issuance to be implemented within 60 days.

Amendment No.: 120

Facility Operating License No. NPF-49. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: May 10, 1995 (60 FR 24912) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 18, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50-278, Peach Bottom Atomic Power Station, Unit No. 3, York County, Pennsylvania

Date of application for amendment: September 1, 1995

Brief description of amendment: The amendment deleted License Condition 2.C.(5) which restricts power levels to no less than seventy percent in the coastdown condition.

Date of issuance: October 17, 1995

Effective date: As of date of issuance

Amendment No.: 215

Facility Operating License No. (DPR-56): This amendment revised the Facility Operating License. Public comments requested as to proposed no significant hazards consideration: Yes. (60 FR 48530). That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by October 18, 1995, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final no significant hazards consideration determination are contained in a Safety Evaluation dated October 17, 1995.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. Vice President and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education

Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of application for amendments: May 19, 1995

Brief description of amendments: The amendments revise the Technical Specifications Table 3.3.3-3, ''Emergency Core Cooling System Response Times'' to reflect the value of 60 seconds for the High Pressure Coolant Injection system response time instead of 30 seconds as previously specified.

Date of issuance: October 16, 1995

Effective date: For both units, as of the date of issuance and to be implemented within 30 days.

Amendment Nos.: 102 and 66

Facility Operating License Nos. NPF-39 and NPF-85. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35084) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 16, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464

Power Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of application for amendment: July 21, 1995

Brief description of amendment: The amendment revises TS Section 6.0 (Administrative Controls) to replace the title-specific list of members on the Plant Operating Review Committee with a more general statement of membership requirements, and expands the scope of disciplines represented on the committee to include Nuclear Licensing and Quality Assurance. The amendment also changes the following management position titles: ''First Executive Vice President and Chief Nuclear Officer'' to ''Chief Nuclear Officer'', ''Resident Manager'' to ''Site Executive Officer'', ''Shift Supervisor'' to ''Shift Manager'', and ''Assistant Shift Supervisor'' to ''Control Room Supervisor.'' These changes in title do not affect the reporting relationships, authority, or responsibilities of these positions. Finally, the amendment also makes editorial corrections to the TSs.

Date of issuance: October 13, 1995

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 228

Facility Operating License No. DPR-59: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: September 13, 1995 (60 FR 47624) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 13, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126

Power Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of application for amendment: April 12, 1995.

Brief description of amendment: The amendment extends the surveillance test intervals for the nuclear steam supply system to support 24-month operating cycles. Surveillance test interval extensions that are justified will be denoted as being performed ''every 24 months'' or ''at least once per 24 months'' consistent with the guidance provided in Reference 1. Other surveillances currently performed ''once each operating cycle,'' ''at least once during each operating cycle,'' ''each refueling,'' or similar notation, that are not being extended at this time will be denoted as being performed ''at least once per 18 months.'' The NRC staff has determined that the proposed TS changes follow the guidance of Generic Letter 91-04, and are therefore acceptable.

Date of issuance: October 13, 1995

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 229

Facility Operating License No. DPR-59: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: May 10, 1995 (60 FR 24916) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 13, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: April 18, 1995

Brief description of amendment: This amendment changes Technical Specification Table 4.3.7.1-1, ''Radiation Monitoring Instrumentation Surveillance Requirements,'' to increase the channel functional test interval from monthly to quarterly for each instrument.

Date of issuance: October 16, 1995

Effective date: As of the date of issuance, to be implemented within 60 days.

Amendment No.: 83

Facility Operating License No. NPF-57: This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 16, 1995 (60 FR 42607) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 16, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: May 4, 1995

Brief description of amendment: This amendment changes Technical Specification (TS) 3/4.6.1.8, ''Drywell and Suppression Chamber Purge System,'' increasing the annual operational limit for the drywell and suppression chamber purge system from 120 to 500 hours.

Date of issuance: October 16, 1995

Effective date: As of the date of issuance, to be implemented within 60 days.

Amendment No.: 84

Facility Operating License No. NPF-57: This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 16, 1995 (60 FR 42607) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 16, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of applications for amendment: November 30, 1994 and March 30, 1995, as supplemented by letter dated September 5, 1995.

Brief description of amendment: The change to TS Table 3.3.1-2, ''Reactor Protection System Response Times,'' TS Table 3.3.2-3, ''Isolation System Instrumentation Response Time,'' TS Table 3.3.3-3, ''Emergency Core Cooling System Response Times,'' and associated Bases, eliminates the requirement to perform response time testing for certain classes of equipment and transfers the requirements of the above-referenced TS Tables to the Updated Final Safety Analysis Report.

Date of issuance: October 24,1995

Effective date: As of date of issuance, to be implemented within 60 days.

Amendment No.: 85

Facility Operating License No. NPF-57: This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: March 29, 1995 (60 FR 16198 and August 16, 1995 (60 FR 42606) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 24, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of application for amendment: May 20, 1994, as supplemented on March 29, 1995

Brief description of amendment: The amendment revises Technical Specifications to implement the NRC's Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactor by relocating specifications that do not meet policy statement criteria to the Final Safety Analysis Report.

Date of issuance: October 20, 1995

Effective date: Immediately, to be implemented within 120 days.

Amendment No.: 103

Facility Operating License No. NPF-30. Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: August 31, 1994 (59 FR 45036). The March 29, 1995, letter provided supplemental information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original Federal Register notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 20, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251

Wisconsin Electric Power Company, Docket Nos. 50-266 and 50-301, Point Beach Nuclear Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of application for amendments: January 26, 1994, as supplemented by letters dated December 1, 1994, and June 23, 1995

Brief description of amendments: These amendments revise Technical Specification (TS) Section 15.3.0, ''General Considerations.'' This section specifies the actions to be taken for conditions not directly addressed in the action statements fo the TSs. In addition, changes to the applicable bases (including the bases for TS 15.3.3) and editorial changes are also included.

Date of issuance: October 12, 1995

Effective date: October 12, 1995

Amendment Nos.: 163 and 167

Facility Operating License Nos. DPR-24 and DPR-27: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: March 16, 1994 (59 FR 12373) The December 1, 1994 and June 23, 1995, submittals provided supplemental information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 12, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241

Wisconsin Electric Power Company, Docket Nos. 50-266 and 50-301, Point Beach Nuclear Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of application for amendments: April 17, 1995

Brief description of amendments: These amendments change TS Sections 15.6.2, ''Organization,'' and 15.6.3, ''Facility Staff Qualifications.'' The requirement for the Operations Manager to hold an NRC Senior Reactor Operator's (SRO) license has been changed to provide additional staffing flexibility.

Date of issuance: October 12, 1995

Effective date: October 12, 1995

Amendment Nos.: 164 and 168

Facility Operating License Nos. DPR-24 and DPR-27. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: May 23, 1995 (60 FR 27346). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 12, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241

Dated at Rockville, Maryland, this 1st day of November 1995.

For the Nuclear Regulatory Commission

Jack W. Roe,

Director, Division of Reactor Projects - III/IV,Office of Nuclear Reactor Regulation

[Doc. 95–27543 Filed 11–7–95; 8:45 am]

BILLING CODE 7590–01–F

[Docket Nos. 50–390 and 50–391]

Tennessee Valley Authority; Availability of Safety Evaluation Report Supplement Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2

The U.S. Nuclear Regulatory Commission has published the Safety Evaluation Report, Supplement 18 (NUREG–0847, Supp. 18) related to the operation of Watts Bar Nuclear Plant, Units 1 and 2, Docket Nos. 50–390 and 50–391.

Copies of the report have been placed in the NRC's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, D.C. 20555, and in the Local Public Document Room, Chattanooga-Hamilton Library, 1001 Broad Street, Chattanooga, Tennessee 37402, for review by interested persons. Copies of the report may be purchased from the Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, D.C. 20013–7082. GPO deposit account holders may charge orders by calling 202–512–2249 or 2171. Copies are also available from the National Technical Information Service, 5285 Port Royal Road, Springfield, Virginia 22161.

Dated at Rockville, Maryland this 31st day of October 1995.